Northeast
Nuclear Energy

Rope Ferry Rd. (Route 156), Waterford, CT 06385

Millstone Nuclear Power Station
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385-0128
(860) 447-1791
Fax (860) 444-4277

The Northeast Utilities System

MAR 10 2000

Docket No. 50-336
B18018

Re: 10 CFR 50.73(a)(2)(iv)

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Nuclear Power Station, Unit No. 2
Licensee Event Report 2000-003-00
Manual Reactor Trip After Two Control Element Assemblies
Unexpectedly Drop into the Core

This letter forwards Licensee Event Report (LER) 2000-003-00, documenting an event that occurred at Millstone Nuclear Power Station, Unit No. 2, on February 11, 2000. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(iv).

The Northeast Nuclear Energy Company's (NNECO's) commitments contained in this letter are located in Attachment 1.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

FOR: C. J. Schwarz
Station Director

BY: ______________________
D. S. McCracken
Assistant Station Director - Safety

Attachments (2): List of Regulatory Commitments
LER 2000-003-00

cc: H. J. Miller, Region I Administrator
J. I. Zimmerman, NRC Project Manager, Millstone Unit No. 2
D. P. Beaulieu, Senior Resident Inspector, Millstone Unit No. 2

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Attachment 1

Millstone Nuclear Power Station, Unit No. 2

List of Regulatory Commitments

March 2000

List of Regulatory Commitments

The following table identifies those actions committed to by NNECO in this document.

Number	Commitment	Due
B18018-01	Establish a preventive maintenance program for periodic refurbishment of power switch modules, to include guidance for bench testing power switches and adjusting voltage output under load, and for any other periodic maintenance deemed appropriate.	September 1, 2000

Docket No. 50-336
B18018

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

LER 2000-003-00

March 2000

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 2	05000336	1 OF 4

TITLE (4)

Manual Reactor Trip After Two Control Element Assemblies Unexpectedly Drop into the Core

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
02	11	2000	2000	– 003 –	00	03	10	2000	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	096		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
R. Joshi, MP2 Acting Regulatory Compliance Supervisor	(860) 440-2080

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On February 11, 2000 at approximately 1346 hours with the unit in Mode 1 at 96 percent power, a manual reactor trip was initiated during Control Element Assembly (CEA) insertion testing after the second of two CEAs unexpectedly dropped into the core. Immediately prior to the trip, Operators were reducing power in accordance with Technical Specification Action Statement 3.1.3.1(e) after the first CEA had dropped into the core. During this power reduction action, the second CEA dropped into the core approximately 17 minutes after the first.

As directed by procedure, the Operators then manually tripped the reactor immediately after the second CEA inserted at which time all remaining CEAs inserted completely into the core. The Auxiliary Feedwater System automatically initiated as a result of the trip and other appropriate safety systems operated as expected. The plant was stabilized in hot standby (Mode 3) at normal post trip parameters and Operating shift personnel reacted appropriately throughout the trip event.

The apparent cause of the first CEA failure was determined to be a ground in the circuit between the interposing cabling and the coil stack. In addition, the apparent cause of the second CEA failure was determined to be failed relays in the assembly's upper gripper circuit. As a contributing factor, it was determined that there were deficiencies in the CEA's preventive maintenance program.

Prior to unit restart, several corrective actions were performed to both test and replace suspect equipment. In addition, a preventive maintenance program for periodic refurbishment of power switch modules will be established by September 1, 2000.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	2000	- 003 -	00	2 OF 4

TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

I. Description of Event

On February 11, 2000 at approximately 1346 hours with the unit in Mode 1 at 96 percent power, a manual reactor [RCT] trip was initiated during Control Element Assembly (CEA) insertion testing after the second of two CEAs unexpectedly dropped into the core. Immediately prior to the trip, Operators were reducing power in accordance with Technical Specification Action Statement 3.1.3.1(e) after the first CEA had dropped into the core. During this power reduction action, the second CEA dropped into the core approximately 17 minutes after the first. As directed by procedure, the Operators then manually tripped the reactor immediately after the second CEA inserted at which time all remaining CEAs inserted completely into the core.

The Auxiliary feedwater System [BA] automatically initiated as a result of the trip and other appropriate safety systems operated as expected. The plant was stabilized in hot standby (Mode 3) at normal post trip parameters and operating shift personnel reacted appropriately throughout the trip event.

Because this event resulted in both a manual actuation of the Reactor Protection System [JC] and an automatic actuation of Engineered Safety Feature [JE] equipment, it is being reported pursuant to 10CFR50.73(a)(2)(iv).

II. Cause of Event

The apparent cause of the first CEA failure (#7-65) was determined to be a ground in the circuit between the interposing cabling [CBL] and the coil [CL] stack. In addition, the apparent cause of the second CEA failure (#3-63) was determined to be failed relays [RLY] in the assembly's upper gripper circuit. As a contributing factor, it was determined that there were deficiencies in the CEA's preventive maintenance program.

III. Analysis of Event

The purpose of the CEAs and associated control element drive system (CEDS) [AA] are: to provide rapid insertion of the CEAs into the reactor core to shutdown the reactor; to move the CEAs in a controlled manner for reactivity changes and maintaining the specified rod position by holding the CEAs in a fixed position. The CEAs provide a means for controlling the reactivity of the reactor core such that specified acceptable fuel design limits are not exceeded. The CEAs are attached to and raised or lowered by control element drive mechanisms (CEDMs). The CEDM is an electromagnetic jack device. It moves, holds or releases its CEA in response to signals from its power cabinet [CAB]. For a reactor trip, the reactor trip breakers [BKR] are open and interrupt the signal from the power cabinets to the CEDMs causing the CEAs to drop into the core.

Dropped CEA events are analyzed in FSAR Chapter 14, Section 4.3.1. As noted in the FSAR, a control rod/bank drop is initiated by the failure of the CEDM or another failure in the CEDS. Upon rod insertion reactor power initially decreases due to negative reactivity insertion. Power will then recover due to feedback from moderator and Doppler reactivity. Changes in local peaking will occur due to the local effect of the dropped rod. The bounding case is that of a dropped bank of CEAs.

An automatic reactor trip is typically expected for a dropped CEA bank, while no trip is expected for a single dropped CEA. The specific event documented in this LER, a dropped CEA followed by a power reduction followed by a second dropped CEA in different bank of CEAs, is not specifically analyzed. Post-trip review of the second dropped CEA has shown that the FSAR analysis bounds this event. Since appropriate safety equipment functioned as expected, and the health and safety of the public were not affected, this event is considered to be of low safety significance.

IV. Corrective Action

Following the reactor trip and prior to restarting the unit, the following remedial corrective actions were performed:

1. The CEA 7-65 coil stack, upper gripper power switch [JS], and pull down coil power switch were replaced.
2. For CEA 3-63, the upper gripper power switch was replaced.
3. To reduce aging/cyclic related fuse [FU] failures, all of the CEAs 15 amp fuses were replaced with 30 amp fuses.
4. Bench testing was performed on each power switch to ensure acceptable contact resistance of the reed relays and resistance checks of the silicon controlled rectifiers (SCRs) [RECT]. Components that did not meet acceptance criteria were replaced.
5. Current traces were conducted on all of the CEAs and compared with ideal traces.

Additionally, the following corrective action will be performed:

1. By September 1, 2000, establish a preventive maintenance program for periodic refurbishment of power switch modules, to include guidance for bench testing power switches and adjusting voltage output under load, and for any other periodic maintenance deemed appropriate.

Discussions with the equipment vendor (Combustion Engineering) resulted in several recommended actions that could increase system reliability. These as well as other corrective actions are being evaluated and addressed via the Millstone Corrective Action Program.

V. Additional Information

Similar Events

As a pending corrective action for the first similar event (LER 99-012), the CEA 7-65 coil stack was to be examined and replaced in the upcoming refueling outage 13. As described in the cause of event for this LER, a potential failure mechanism for this CEA involved this component. As a result, if the coil stack had been replaced following the 99-012 event, CEA 7-65's present failure may have been avoided.

LER 99-012: This LER identified an event where Technical Specification Action Statement (TSAS) 3.0.3 was invoked (the first time) when a Control Element Assembly (CEA 7-65) could not be aligned with its group within one hour as required by Technical Specifications (TSAS 3.1.3.1.e, Movable Control Element Assembly, Full Length CEA Position). During the shutdown process the TOTAL UNRODDED INTEGRATED RADIAL PEAKING FACTOR (F_r^T) exceeded its 100% power limit while the TSAS for Power Distribution Limits, AZIMUTHAL POWER TILT (T_q) (TSAS 3.2.4.a) was in effect. This condition also required invoking TSAS 3.0.3 (the second time). The cause of this event was a grounded coil wire for the lower gripper assembly. The reason for the ground cannot be determined until the coil stack is analyzed. F_r^T and T_q exceeded TS limits due to local power peaking effects caused by the misaligned CEA and subsequent reactor power reduction. As corrective actions, the leads leading to lower gripper coil for the CEA were insulated, and the

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Millstone Nuclear Power Station Unit 2	05000336	2000 --	003 --	00	4 OF 4

TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

coil stack for the CEA were planned to be replaced during refueling outage 13 and examined for cause of failure of the insulator.

LER 94-009: This LER identified an event in which CEA 7-65 was declared immovable and it was identified that the Technical Specification action statement requirements for an immovable CEA had not been performed within the specified time. Power was removed to the Control Element Drive Mechanism (CEDM) for CEA 7-65 and the CEA fully inserted into the core and the reactor was manually tripped. The cause of the failure of CEA 7-65 to insert upon demand was determined to be an intermittent failure of one (of three) silicon controlled rectifier (SCR) in the upper gripper coil power switch. Corrective actions for this event included:

1. Replacing the upper gripper power switch module for CEA 7-65.
2. Revising procedure AOP 2556, "CEA Malfunctions" to provide guidance for each of the Technical Specification action statement requirements for CEA position and CEA position indicator channels.

LER 95-031: This LER identified an event in which a plant shutdown was commenced, in accordance with TSAS 3.0.3, when it was determined that CEA 1-30 could not be aligned with its group within one hour as required by TSAS 3.1.3.1.e. The cause of the event was a failed Coil Power Programmer (CPP) power supply due to a fabrication error. A drop of solder across a capacitor created a short circuit, resulting in a the loss of the power supply unit. Corrective actions of this event included replacing the defective power supply unit and visual inspection of thirty-five additional power supplies. No additional defects were found.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].